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New York Power Authority

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August 5, 1993
IPN-93-094

U.S. Nuclear Regulatory Commission
Mail Station P1-137
Washington, DC 20555

Attn: Document Control Desk

Subject: **Indian Point 3 Nuclear Power Plant**
Docket No. 50-286
Response to Generic Letter 93-04
Rod Control System Failure and
Withdrawal of Rod Control Cluster Assemblies

References:

1. NRC Generic Letter 93-04, "Rod Control System Failure and Withdrawal of Rod Control Cluster Assemblies, 10 CFR 50.54 (f)", dated June 21, 1993.

2. Letter from Ashok C. Thadani (Director, Division of Systems and Safety and Analysis, Office of Nuclear Reactor Regulation, Nuclear Regulatory Commission) to Roger Newton (Chairman, Westinghouse Owners Group, Regulatory Response Group), dated July 26, 1993, "WOG Request for Schedular Relief in Responding to NRC Generic Letter 93-04".

Dear Sir:

This letter provides the Authority's response to Generic Letter (GL) 93-04, "Rod Control System Failure and Withdrawal of Rod Control Cluster Assemblies," dated June 21, 1993, for the Indian Point 3 Nuclear Power Plant. This generic letter was addressed to all licensees with the Westinghouse Rod Control System for action and to all other licensees for information.

The generic letter required each licensee with a Westinghouse Rod Control System to submit a written response, per 10 CFR 50.54 (f), within 45 days from the date of the generic letter. This response was to provide an assessment of whether or not the licensing basis for each facility was satisfied with regard to the requirements for system response to a single failure in the Rod Control System, as stipulated in General Design Criteria (GDC 25), in light of the information

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generated as a result of the Salem event. If the assessment indicated that the licensing basis was not satisfied, the response was to also include an assessment of the impact of potential single failures in the rod control system on the licensing basis of the facility and a description of compensatory short-term actions to address any actual or potential degraded or nonconforming conditions. If the assessment indicated that the licensing basis was not satisfied, the licensee was to provide a plan and schedule for the long-term resolution of this issue within 90 days from the date of the generic letter.

On July 9, 1993 representatives of the Westinghouse Owner's Group (WOG), including a representative from the Authority, met with the NRC staff to discuss the WOG's strategy to resolve the rod control system issue and the response to GL 93-04. As discussed at that meeting, the WOG's plan consists of two programs: (1) the rod control system evaluation program to assess the historical performance of the rod control system and to determine the type of rod motion that can occur when the drive mechanisms receive incorrect orders and (2) the safety analysis program to show compliance with GDC 25. The Authority has supported these efforts and has provided the WOG with information on Indian Point 3's rod control system history. The Authority received the Westinghouse report, "Summary of the Generic Safety Analysis Program", on Friday, July 30, 1993 and the final site specific analysis results on Wednesday, August 4, 1993.

As stated in Reference 2, to allow adequate time for licensees to review these reports, the NRC staff has agreed that the 45 day response need only describe the short term actions being taken, and the results from the generic safety analysis program and its applicability to individual licensees. The assessment of whether or not the licensing basis for each facility is satisfied with regard to the requirements for system response to a single failure in the rod control system may be deferred to the 90 day response to the generic letter.

As such, this submittal contains the following information. Attachment I of this letter provides a description of the short term actions being taken at Indian Point 3. The Authority's 90 day response will address the licensing basis issue. Attachment II provides Westinghouse report, "Summary of the Generic Safety Analysis Program" and the Authority's preliminary assessment of the applicability of the generic safety analysis program to Indian Point 3. Within two weeks of receipt of Westinghouse report WCAP-13803, entitled "Generic Assessment of Asymmetric Rod Cluster Control Assembly Withdrawal", the Authority will confirm the applicability of the generic safety analysis program to Indian Point 3. This WCAP will provide an overview of the methodology used to perform this analysis. If the Authority's review of WCAP-13803 determines that the generic safety analysis program is not applicable to Indian Point 3, the Authority will advise the NRC in writing at that time. Attachment III to this letter provides a list of the commitments being made by the Authority with this submittal.

The Authority will implement the short term corrective actions provided as Attachment I to this submittal prior to restart. In addition, the Authority will review the applicability of any additional information regarding this issue to Indian Point 3, as it becomes available, as well as re-evaluating the appropriateness of the actions committed to in this submittal.

If you have any questions regarding this matter, please contact Mr. P. Kokolakis.

Very truly yours,

Ralph E. Beedle

STATE OF NEW YORK
COUNTY OF WESTCHESTER
Subscribed and Sworn to before me
this 5th day of August 1993

Kathleen D. Gallagher
Notary Public

KATHLEEN D. GALLAGHER
Notary Public, State of New York
No. 5004481
Qualified in Westchester County
Commission Expires Nov. 16, 1994

Attachment

cc: See next page

att: as stated

cc: U.S. Nuclear Regulatory Commission
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ATTACHMENT I TO IPN-93-094

SHORT-TERM ACTIONS TO ADDRESS

ACTUAL OR POTENTIAL DEGRADED OR NONCONFORMING CONDITIONS

PERTAINING TO

SINGLE FAILURE VULNERABILITY OF WESTINGHOUSE ROD CONTROL SYSTEM

NEW YORK POWER AUTHORITY
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286
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The purpose of this discussion is to provide a response to the three areas of compensatory short-term actions identified by the NRC in Generic Letter (GL) 93-04, Required Response 1 (b).

1. "additional cautions or modifications to surveillance and preventive maintenance procedures"

The Power Authority has assessed Indian Point 3's control rod drive system's surveillance, operations and preventive maintenance procedures in light of the recent industry events, and has determined that procedural changes are warranted. As such, prior to Indian Point 3's return to power operation, all procedures pertaining to the control rod drive system will be revised, as appropriate, to include additional guidance to operations, maintenance, and instrumentation and control (I&C) personnel. Precautionary measures to verify the correct operation of the system will also be incorporated as necessary. For example, following the performance of maintenance on the system, each bank of RCCAs will be exercised to ensure the correct motion of the rods.

In addition to these procedural enhancements, the Authority will implement several precautionary measures prior to returning Indian Point 3 to power operation. First, the functionality of the rod deviation alarm will be verified prior to restart. Second, each bank of RCCAs will be individually exercised prior to withdrawing the shutdown bank and bringing the reactor critical. Satisfactory performance of these measures will enhance confidence in Indian Point 3's rod control system's ability to perform its intended function.

As required by Technical Specifications, Indian Point 3 conducts an RCCA operability test every 31 days. In addition, Indian Point 3 verifies monthly the functionality of the rod deviation alarms with simulated signals. The Authority believes these surveillance frequencies to be adequate. However, as part of the Authority's ongoing review of new available industry data pertaining to this issue, the Authority will re-evaluate the appropriateness of increasing these surveillance frequencies.

2. "additional administrative controls for plant startup and power operation"

The Power Authority agrees that additional administrative controls for plant startup and power operation are prudent in light of recent industry events. Consequently, as previously stated, the Authority will revise Indian Point 3's surveillance, operations and preventive maintenance procedures to include additional guidance and precautionary measures. These administrative enhancements, combined with the precautionary measures which will be taken prior to returning the unit to operation, as delineated above, will help ensure the safe operation of Indian Point 3.

It is important to note that Indian Point 3 is a base-load plant and does not normally follow load demands. As a result, the rod control system is normally operated with all rods out. Long-term reactivity adjustments are handled by adjusting boron concentration. This operating approach reduces the potential for unplanned rod withdrawal events.

In addition, the Indian Point 3 rod control system is equipped with mechanical step counters, which produce audible clicks in the control room whenever rods move in either direction. This feature provides additional assurance that operators will immediately be alerted to and terminate any rod control system failures which result in an unplanned movement of control rods.

3. "additional instructions and training to heighten operator awareness of potential rod control system failures and to guide operator response in the event of a rod control system malfunction"

Indian Point 3 is taking a number of actions to enhance operator awareness. As part of their routine training, the licensed reactor operators are currently being trained on the rod control system. This training module has been revised to include a discussion of the malfunctions at Salem and appropriate operator response in the event of a rod control system malfunction. Instrumentation and Control personnel and associated systems engineers will be informed of the Salem event prior to restart. In addition, abnormal event response procedures are also being reviewed to assure adequate guidance to operators in the event of a rod control system malfunction. This review will be complete and the procedures revised, if necessary, prior to restart. These measures will not only heighten operator awareness, but will ensure that appropriate guidance is available to help the operators respond appropriately in the event of a rod control system malfunction.

The Power Authority has actively participated in the Westinghouse Owner's Group's (WOG) efforts to address this issue and is committed to continue supporting the WOG's efforts.

ATTACHMENT II TO IPN-93-094

APPLICABILITY OF GENERIC SAFETY ANALYSIS

PERTAINING TO

SINGLE FAILURE VULNERABILITY OF WESTINGHOUSE ROD CONTROL SYSTEM

1. **The Power Authority's Preliminary Assessment of the Applicability of the Westinghouse Generic Safety Analysis to Indian Point 3**

2. **Westinghouse Report, "Summary of the Generic Safety Analysis Program"**

NEW YORK POWER AUTHORITY
INDIAN POINT 3 NUCLEAR POWER PLANT
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DPR-64

The Power Authority's Preliminary Assessment of the Applicability of the Westinghouse Generic Safety Analysis to Indian Point 3

The Authority has reviewed Westinghouse's report entitled, "Summary of the Generic Safety Analysis Program", which is included in this attachment. The Authority reviewed the Indian Point 3 specific data used in this analysis and determined that it was representative of Indian Point 3. In addition, the Authority has reviewed the initial site specific results of this analysis for Indian Point 3, and anticipates that the results presented in the final version will show that the generic safety analysis program is applicable to Indian Point 3. As such, Departure from Nucleate Boiling (DNB) is not anticipated to occur for the postulated worst-case asymmetric rod withdrawal event. However, as stated in the cover letter of this submittal, the Authority will review WCAP-13803, entitled "Generic Assessment of Asymmetric Rod Cluster Control Assembly Withdrawal", to confirm the applicability of the generic analysis. This WCAP will provide an overview of the methodology used in this analysis.

Summary of the Generic Safety Analysis Program

Introduction

As part of the Westinghouse Owners Group initiative, the WOG Analysis subcommittee is working on a generic approach to demonstrate that for all Westinghouse plants there is no safety significance for an asymmetric RCCA withdrawal. The purpose of the program is to analyze a series of asymmetric rod withdrawal cases from both subcritical and power conditions to demonstrate that DNB does not occur.

The current Westinghouse analysis methodology for the bank withdrawal at power and from subcritical uses point-kinetics and one dimensional kinetics transient models, respectively. These models use conservative constant reactivity feedback assumptions which result in an overly conservative prediction of the core response for these events.

A three-dimensional spatial kinetics/systems transient code (LOFT5/SPNOVA) is being used to show that the localized power peaking is not as severe as current codes predict. The 3-D transient analysis approach uses a representative standard 4-Loop Westinghouse plant with conservative reactivity assumptions. Limiting asymmetric rod withdrawal statepoints (i.e., conditions associated with the limiting time in the transient) are established for the representative plant which can be applied to all Westinghouse plants. Differences in plant designs are addressed by using conservative adjustment factors to make a plant-specific DNB assessment.

Description of Asymmetric Rod Withdrawal

The accidental withdrawal of one or more RCCAs from the core is assumed to occur which results in an increase in the core power level and the reactor coolant temperature and pressure. If the reactivity worth of the withdrawn rods is sufficient, the reactor power and/or temperature may increase to the point that the transient is automatically terminated by a reactor trip on a High Nuclear Flux or Over-Temperature Delta-T (OTDT) protection signal. If the reactivity rise is small, the reactor power will reach a peak value and then decrease due to the negative feedback effect caused by the moderator temperature rise. The accidental withdrawal of a bank or banks of RCCAs in the normal overlap mode is a transient which is specifically considered in plant safety analysis reports. The consequences of a bank withdrawal accident meet Condition II criteria (no DNB). If, however, it is assumed that less than a full group or bank of control rods is withdrawn, and these rods are not symmetrically located around the core, this can cause a "tilt" in the core radial power distribution. The "tilt" could result in a radial power distribution peaking factor which is more severe than is normally considered in the plant safety analysis report, and therefore cause a loss of DNB margin. Due to the imperfect mixing of the fluid exiting the core before it enters the hot legs of the reactor coolant loops, there can be an imbalance in the loop temperatures, and therefore in the measured values of T-avg and delta-T, which are used in the Over-Temperature Delta-T protection system for the core. The radial power "tilt" may also affect the ex-core detector signals used for the High Nuclear Flux trip. The axial offset (AO) in the region of the core where the rods are withdrawn may become more positive than the remainder of the core, which can result in an additional DNB penalty.

Methods

The LOFT5 computer code is used to calculate the plant transient response to an asymmetric rod withdrawal. The LOFT5 code is a combination of an advanced version of the LOFT4 code (Reference 1), which has been used for many years by Westinghouse in the analysis of the RCS behavior to plant transients and accidents, and the advanced nodal code SPNOVA (Reference 2).

LOFT5 uses a full-core model, consisting of 193 fuel assemblies with one node per assembly radially and 20 axial nodes. Several "hot" rods are specified with different input multipliers on the hod rod powers to simulate the effect of plants with different initial $F\Delta H$ values. A "hot" rod represents the fuel rod with the highest $F\Delta H$ in the assembly, and is calculated by SPNOVA within LOFT5. DNBRs are calculated for each hot rod within LOFT5 with a simplified DNB-evaluation model using the WRB-1 correlation. The DNBRs resulting from the LOFT5 calculations are used for comparison purposes.

A more detailed DNBR analysis is done at the limiting transient statepoints from LOFT5 using THINC-IV (Reference 3) and the Revised Thermal Design Procedure (RTDP). RTDP applies to all Westinghouse plants, maximizes DNBR margins, is approved by the NRC, and is licensed for a number of Westinghouse plants. The LOFT5-calculated DNBRs are conservatively low when compared to the THINC-IV results.

Assumptions

The initial power levels chosen for the performance of bank and multiple RCCA withdrawal cases are 100%, 60%, 10% and hot zero power (HZP). These power levels are the same powers considered in the RCCA Bank Withdrawal at Power and Bank Withdrawal from Subcritical events presented in the plant Safety Analysis Reports. The plant, in accordance with RTDP, is assumed to be operating at nominal conditions for each power level examined. Therefore, uncertainties will not affect the results of the LOFT5 transient analyses. For the at-power cases, all reactor coolant pumps are assumed to be in operation. For the hot zero power case (subcritical event), only 2/4 reactor coolant pumps are assumed to be in operation. A "poor mixing" assumption is used for the reactor vessel inlet and outlet mixing model.

Results

A review of the results presented in Reference 4 indicates that for the asymmetric rod withdrawal cases analyzed with the LOFT5 code, the DNB design basis is met. As demonstrated by the A-Factor approach (described below) for addressing various combinations of asymmetric rod withdrawals, the single most-limiting case is plant-specific and is a function of rod insertion limits, rod control pattern, and core design. The results of the A-Factor approach also demonstrates that the cases analyzed with the LOFT5 computer code are sufficiently conservative for a wide range of plant configurations for various asymmetric rod withdrawals. In addition, when the design $F\Delta H$ is taken into account on the representative plant, the DNBR criterion is met for the at-power cases.

At HZP, a worst-case scenario (3-rods withdrawn from three different banks which is not possible) shows a non-limiting DNBR. This result is applicable to all other Westinghouse plants.

Plant Applicability

The 3-D transient analysis approach uses a representative standard 4-Loop Westinghouse plant with bounding reactivity assumptions with respect to the core design. This results in conservative asymmetric rod(s) withdrawal statepoints for the various asymmetric rod withdrawals analyzed. The majority of the cases analyzed either did not generate a reactor trip or were terminated by a High Neutron Flux reactor trip. For the Overtemperature Delta-T reactor trip, no credit is assumed for the f(ΔI) penalty function. The f(ΔI) penalty function reduces the OTDT setpoint for highly skewed positive or negative axial power shapes. Compared to the plant-specific OTDT setpoints including credit for the f(ΔI) penalty function, the setpoint used in the LOFT5 analyses is conservative, i.e., for those cases that tripped on OTDT, a plant-specific OTDT setpoint with the f(ΔI) penalty function will result in an earlier reactor trip than the LOFT5 setpoint. This ensures that the statepoints generated for those cases that trip on OTDT are conservative for all Westinghouse plants.

With respect to the neutronic analyses, an adjustment factor ("A-factor") was calculated for a wide range of plant types and rod control configurations. The A-factor is defined as the ratio between the design FΔH and the change in the maximum transient FΔH from the symmetric and asymmetric RCCA withdrawal cases. An appropriate and conservative plant-specific A-factor was calculated and used to determine the corresponding DNBR penalty or benefit. With respect to the thermal-hydraulic analyses, differences in plant conditions (including power level, RCS temperature, pressure, and flow) are addressed by sensitivities performed using THINC-IV. These sensitivities are used to determine additional DNBR penalties or benefits. Uncertainties in the initial conditions are accounted for in the DNB design limit. Once the differences in plant design were accounted for by the adjustment approach, plant-specific DNBR calculations can be generated for all Westinghouse plants.

Conclusion

Using this approach, the generic analyses and their plant-specific application demonstrate that for (Plant Name) DNB does not occur for their worst-case asymmetric rod withdrawal.

References

1) Burnett, T.W.T., et al., "LOFTRAN Code Description," WCAP-7907-A, April 1984.

2) Chao, Y.A., et al., "SPNOVA - A Multi-Dimensional Static and Transient Computer Program for PWR Core Analysis," WCAP-12394, September 1989.

3) Friedland, A.J. and S. Ray, "Improved THINC IV Modeling for PWR Core Design," WCAP-12330-P, August 1989.

4) Huegel, D., et al., "Generic Assessment of Asymmetric Rod Cluster Control Assembly Withdrawal," WCAP-13803, August 1993.

ATTACHMENT III TO IPN-93-094

AUTHORITY COMMITMENTS

RELATED TO

SINGLE FAILURE VULNERABILITY OF WESTINGHOUSE ROD CONTROL SYSTEM

NEW YORK POWER AUTHORITY
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286
DPR-64

Commitment Number	Commitment	Due Date
93-094-01	Submit 90 day response addressing GDC 25	September 20, 1993
93-094-02	Review applicability of any additional information and re-evaluate appropriateness of commitments made in IPN-93-094.	To be determined if additional information is received.
93-094-03	Revise procedures pertaining to the rod control system to include additional guidance to operations, maintenance and I&C personnel.	Prior to restart
93-094-04	Revise procedures pertaining to the rod control system to include precautionary measures to verify the correct operation of the system.	Prior to restart
93-094-05	Revise procedures to require the exercising of each bank of RCCAs to ensure the correct motion of the rods following the performance of maintenance on the control rod drive system.	Prior to restart
93-094-06	Verify functionality of rod deviation alarm with simulated signals.	Prior to restart
93-094-07	Verify functionality of the rod deviation alarms with simulated signals monthly.	This is a current procedural requirement
93-094-08	Perform RCCA operability test every 31 days.	This is a current procedural requirement
93-094-09	Exercise each RCCA bank individually prior to withdrawing the shutdown bank and bringing the reactor critical.	Prior to restart
93-094-10	Train licensed reactor operators on rod control system, including a discussion of the malfunctions at Salem and appropriate operator response in the event of a rod control system malfunction.	Prior to restart
93-094-11	Inform I&C staff and associated systems engineers of Salem event.	Prior to restart

93-094-12	Review and revise, as necessary, abnormal event response procedures to assure adequate guidance to operators in the event of a rod control system malfunction.	Prior to restart
93-094-13	Review WCAP-13803 and applicability of generic analysis to Indian Point 3.	Within two weeks of obtaining WCAP-13803